

c. 2

Neutron Production from ( $\alpha, n$ ) Reactions and  
Spontaneous Fission in  $\text{ThO}_2$ ,  $\text{UO}_2$ , and  
 $(\text{U, Pu})\text{O}_2$  Fuels

**DO NOT CIRCULATE**

**PERMANENT RETENTION**

**REQUIRED BY CONTRACT**

University of California



**LOS ALAMOS SCIENTIFIC LABORATORY**

Post Office Box 1663 Los Alamos, New Mexico 87545

This work was supported by the Electric Power Research Institute and the US Department of Energy, Spent Fuel Project Office, under the technical direction of the Savannah River Laboratory.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

UNITED STATES  
DEPARTMENT OF ENERGY  
CONTRACT W-7405-ENG. 36

LA-8869-MS

UC-34c

Issued: June 1981

# Neutron Production from $(\alpha, n)$ Reactions and Spontaneous Fission in $\text{ThO}_2$ , $\text{UO}_2$ , and $(\text{U, Pu})\text{O}_2$ Fuels



R. T. Perry\*  
W. B. Wilson



\*LASL Consultant. University of Wisconsin, 329 Engineering Research Building, Madison, WI 53706.



NEUTRON PRODUCTION FROM ( $\alpha$ ,n) REACTIONS AND SPONTANEOUS FISSION  
IN ThO<sub>2</sub>, UO<sub>2</sub>, AND (U,Pu)O<sub>2</sub> FUELS

by

R. T. Perry and W. B. Wilson

**ABSTRACT**

Available alpha-particle stopping cross-section and <sup>17,18</sup>O( $\alpha$ ,n) cross-section data were adjusted, fitted, and used in calculating the thick-target neutron production function for alpha particles below 10 MeV in oxide fuels. The spent UO<sub>2</sub> function produced was folded with actinide decay spectra to determine ( $\alpha$ ,n) neutron production by each of 89 actinides. Spontaneous-fission (SF) neutron production for 40 actinides was calculated as the product of  $\bar{\nu}$ (SF) and SF branching-fraction values accumulated or estimated from available data. These contributions and total neutron production in spent UO<sub>2</sub> fuel are tabulated and, when combined with any calculated inventory, describe the spent UO<sub>2</sub> neutron source. All data are tabulated and methodology is described to permit easy extension to specialized problems.

---

**I. INTRODUCTION**

Neutron sources are present in reactor fuel from the spontaneous-fission (SF) decay of actinide nuclides and from the interaction of their decay alpha particles with low- and medium-Z nuclides in ( $\alpha$ ,n) reactions. The ( $\alpha$ ,n) source in oxide fuels is dominated by reactions with <sup>17</sup>O and <sup>18</sup>O, which are present in NAT<sub>O</sub> in 0.038 and 0.204 atom percent abundancies, respectively.

The probability of neutron production by an alpha particle emitted at energy  $E_\alpha$  in the fuel is given by the thick-target neutron production function  $P(E_\alpha)$ , which we have evaluated for four fuel compositions--clean ThO<sub>2</sub> thermal

reactor fuel, clean and spent  $UO_2$  thermal reactor fuel, and clean  $(U,Pu)O_2$  fast reactor fuel. The  $(\alpha,n)$  neutron production function has been evaluated at the Hanford Engineering Development Laboratory (HEDL) by Ombrellaro and Johnson for alpha particles in FFTF fuel;<sup>1</sup> however,  $P(E_\alpha)$  has not been calculated for the fuels of interest here, and the change in  $P(E_\alpha)$  with exposure has not been evaluated. We have employed the methodology and data used in the HEDL work<sup>1</sup> with minor exceptions in data and energy range of calculation.

The equations describing  $(\alpha,n)$  and SF neutron production and the data quantities used in the calculations are given in Sec. II. The available data sources and adjustments made to the data are described in Sec. III. Details of the  $(\alpha,n)$  calculations are briefly discussed in Sec. IV. Resulting  $(\alpha,n)$ , SF, and total neutron production values are given in Sec. V for each of a variety of actinide nuclides produced in reactor fuels.

Selected results of these calculations have been reported previously.<sup>2-5</sup>

## II. THEORY

The slowing and stopping of alpha particles in a material are described by the material's alpha-particle stopping power,

$$SP(E) = - \frac{dE}{dx} , \quad (1)$$

which gives the energy-dependent energy loss of alpha particles of energy  $E$  per unit path length  $x$ .<sup>6</sup> The energy loss of an alpha particle of initial energy  $E_\alpha$  in traveling a distance  $X$  can be determined from the stopping power as

$$\Delta E = E_\alpha - E'_\alpha = \int_0^X \left( - \frac{dE}{dx} \right) dx . \quad (2)$$

Similarly, the distance traveled in slowing from  $E_\alpha$  to  $E'_\alpha$  is

$$X = \int_{E'_\alpha}^{E_\alpha} \frac{1}{\left( \frac{dE}{dx} \right)} dE = \int_{E'_\alpha}^{E_\alpha} \frac{1}{\left( - \frac{dE}{dx} \right)} dE . \quad (3)$$

Neutrons may be produced within the material by  $(\alpha, n)$  reactions with nuclide  $i$ , which has atom density  $N_i$  and microscopic  $(\alpha, n)$  cross section  $\sigma_i(E)$ . The probability of  $(\alpha, n)$  interaction with nuclide  $i$  by an alpha particle of energy  $E$  traveling from  $x$  to  $x + dx$  is

$$N_i \sigma_i(E) dx = \frac{N_i \sigma_i(E) dE}{\left(\frac{dE}{dx}\right)} \quad (4)$$

The probability of  $(\alpha, n)$  interaction with nuclide  $i$  by an alpha particle in lieu of slowing from  $E_\alpha$  to  $E'_\alpha$  is then

$$P_i(E_\alpha, E'_\alpha) = \int_{E'_\alpha}^{E_\alpha} \frac{N_i \sigma_i(E) dE}{\left(\frac{dE}{dx}\right)} = \int_{E'_\alpha}^{E_\alpha} \frac{N_i \sigma_i(E) dE}{\left(-\frac{dE}{dx}\right)} \quad (5)$$

The probability of  $(\alpha, n)$  interaction with nuclide  $i$  by an alpha particle prior to stopping in the material is given by the thick-target neutron production function

$$P_i(E_\alpha) = \int_0^{E_\alpha} \frac{N_i \sigma_i(E) dE}{\left(-\frac{dE}{dx}\right)} \quad (6)$$

In addition to that of the above definition of Eq. (1), a variety of quantities are referred to as "stopping powers" or often alternately "stopping cross sections." These include (typically without explicit regard to sign) the quantities  $\frac{dE}{dx} = \frac{dE}{d(\rho x)} = \frac{dE}{\rho dx}$ ,  $\frac{dE}{Z^2 dx}$ ,  $\rho \frac{dE}{dx}$ , and  $\frac{dE}{N dx}$ . Here  $x$  is material thickness ( $\text{mg}/\text{cm}^2$ ),  $Z$  is atomic number,  $\rho$  is material density ( $\text{g}/\text{cm}^3$ ), and  $N$  is the total atom density of the material ( $\text{atoms}/\text{cm}^3$ ). The last quantity is also called the stopping cross section,

$$\epsilon(E) = -\frac{1}{N} \frac{dE}{dx} \quad (7)$$

a notation adopted here. Equations above defining  $p_i$  and  $P_i$  may now be written in terms of  $\epsilon$  as

$$P_i(E_\alpha, E'_\alpha) = \frac{N_i}{N} \int_{E'_\alpha}^{E_\alpha} \frac{\sigma_i(E)}{\epsilon(E)} dE \quad (8)$$

and

$$P_i(E_\alpha) = \frac{N_i}{N} \int_0^{E_\alpha} \frac{\sigma_i(E)}{\epsilon(E)} dE \quad (9)$$

Note that  $p_i$  and  $P_i$  are related by

$$P_i(E_\alpha, E'_\alpha) = P_i(E_\alpha) - P_i(E'_\alpha) \quad (10)$$

The stopping cross section  $\epsilon(E)$  of a material composed of J elemental constituents may be calculated using the Bragg-Kleeman<sup>10</sup> relationship, which may be written as

$$\epsilon(E) \approx \frac{1}{N} \sum_{j=1}^J N_j \epsilon_j(E) \quad (11)$$

where

$$N = \sum_{j=1}^J N_j \quad (12)$$

The accuracy of the approximation of Eq. (11) will be discussed in Sec. III.

A fraction of the decays of nuclide k within the material may be by alpha-particle emission. This fraction  $F_k^\alpha$  of alpha decays may occur with the emission of one of L possible alpha-particle energies. The intensity  $f_{k\ell}^\alpha$  is the fraction of all decays of nuclide k resulting in an alpha particle of energy  $E_{k\ell}$ , and

$$F_k^\alpha = \sum_{\ell=1}^L f_{k\ell}^\alpha \quad (13)$$

The fraction of nuclide k decays resulting in  $(\alpha, n)$  neutron production in a thick-target material containing I nuclides with  $(\alpha, n)$  cross sections is thus

$$R_k(\alpha, n) = \sum_{\ell=1}^L f_{k\ell}^{\alpha} \sum_{i=1}^I P_i(E_{k\ell}) \quad . \quad (14)$$

The SF of an actinide nuclide  $k$  is accompanied by the emission of an average  $\bar{\nu}_k$  (SF) neutrons. The SF activity  $A_k^{SF}$  of nuclide  $k$ , having atom density  $N_k$ , is

$$A_k^{SF} = \lambda_k^{SF} N_k \quad . \quad (15)$$

Here,  $\lambda_k^{SF}$  is the SF decay constant defined by

$$\lambda_k^{SF} = \ln 2 / T_{1/2}^k(SF) \quad , \quad (16)$$

where  $T_{1/2}^k(SF)$  is the SF half-life of nuclide  $k$ . SF is typically only one of  $M$  modes of decay; the total activity due to nuclide  $k$  is

$$A_k = \lambda_k N_k = \sum_{m=1}^M A_k^m \quad , \quad (17)$$

where  $\lambda_k$  is the total decay constant of nuclide  $k$ ,

$$\lambda_k = \sum_{m=1}^M \lambda_k^m = \ln 2 / T_{1/2}^k \quad , \quad (18)$$

and  $T_{1/2}^k$  is the total half-life of nuclide  $k$ . The fraction of nuclide  $k$  decays by SF is given by the SF branching fraction

$$F_k^{SF} = A_k^{SF} / A_k = \lambda_k^{SF} / \lambda_k = T_{1/2}^k / T_{1/2}^k(SF) \quad . \quad (19)$$



The average number of SF neutrons emitted per decay (by any mode) of nuclide k is then

$$R_k(\text{SF}) = F_k^{\text{SF}} \bar{\nu}_k(\text{SF}) \quad . \quad (20)$$

The total number of neutrons, on the average, emitted due to ( $\alpha, n$ ) reactions and SF is

$$R_k = R_k(\alpha, n) + R_k(\text{SF}) \quad . \quad (21)$$

The total neutron source S from ( $\alpha, n$ ) reactions and SF within a material containing K pertinent radionuclides is then

$$S = \sum_{k=1}^K \lambda_k N_k R_k \quad . \quad (22)$$

The evaluation of the quantities  $R_k(\alpha, n)$ ,  $R_k(\text{SF})$ , and  $R_k$  for a number of actinide nuclides is described in the following sections.

### III. DATA

The data quantities required to compute the neutron production fractions  $R_k(\alpha, n)$  and  $R_k(\text{SF})$  for each of the four fuels of interest include the following.

- For each major elemental constituent j of the material:  $N_j$ , the atom density; and  $\epsilon_j(E)$ , the alpha-particle stopping cross section.
- For each nuclide i within the material having an ( $\alpha, n$ ) cross section:  $N_i$ , the atom density; and  $\sigma_i(E)$ , the microscopic ( $\alpha, n$ ) cross section.
- For each nuclide k decaying by alpha decay:  $f_{k\ell}^\alpha$ , the intensity for emission of each L alpha particles; and  $E_{k\ell}$ , the energy of each of L alpha particles.
- For each nuclide k decaying by SF:  $F_k^{\text{SF}}$ , the SF branching fraction; and  $\bar{\nu}_k(\text{SF})$ , the average number of neutrons emitted per SF.

### A. Stopping Cross Section $\epsilon(E)$

Densities of each constituent of each fuel type are given in Table I. The fuel composition of  $UO_2$  LWR fuel is given for clean and spent conditions for the evaluation of the effect of exposure-dependent fuel composition on stopping cross section  $\epsilon$ ; here,  $_{41}Nb$  and  $_{59}Pr$  represent the low- and high-mass fission products, respectively. Concentrations of  $_{93}Np$ ,  $_{95}Am$ , and  $_{96}Cm$  are given for the spent  $UO_2$  fuel, although the minor contributions to  $\epsilon$  from these nuclides are included as plutonium. Elements contributing to the material stopping cross sections are thus O, Nb, Pr, Th, U, and Pu.

A bibliography of experimental and theoretical stopping-power references by Anderson<sup>11</sup> notes that some 900 papers have been published on the subject of ion energy loss in matter. Anderson, noting the observation by Bichsel<sup>12</sup> that stopping powers measured by different groups often did not agree within stated uncertainties, was unable to resolve discrepancies after careful analysis and cautioned that stopping-power data sources should be selected carefully. We have chosen as the major stopping cross-section data source the comprehensive volume edited by Ziegler,<sup>13</sup> which gives tabulated alpha stopping cross-section values and functional fits for elements in the range  $1 \leq Z \leq 92$ .

No values of the alpha-stopping cross section for plutonium were identified, although values for plutonium compounds were found.<sup>7</sup> Northcliffe and Schilling<sup>8</sup> have tabulated values of the stopping power  $dE/d\chi$  for  $Z \leq 92$ . They have shown graphically, for each  $Z$  including  $Z = 94$ , the energy-dependent ratio  $(dE/d\chi)_Z : (dE/d\chi)_{Al}$ . In order to form a stopping cross section for plutonium consistent with the data of Ziegler,<sup>13</sup> we have used the stopping power ratio of Ref. 8 in the expression

$$\epsilon_{Pu} = \epsilon_U \frac{A_{Pu}}{A_U} \left[ (dE/d\chi)_{Pu} : (dE/d\chi)_{Al} \frac{(dE/d\chi)_{Al}}{(dE/d\chi)_U} \right], \quad (23)$$

where all quantities enclosed in brackets [ ] were taken from Ref. 8. Values used and produced in this calculation are given in Table II.

Fourth-degree polynomial functions of the form

$$\ln \epsilon = C_0 + C_1 \ln E + C_2 \ln^2 E + C_3 \ln^3 E + C_4 \ln^4 E \quad (24)$$

TABLE I  
 PROPERTIES OF OXIDE FUELS

	Thermal Reactor Fuels			Fast Reactor Fuel (U,Pu)O <sub>2</sub> Clean
	UO <sub>2</sub> Clean	UO <sub>2</sub> Spent	ThO <sub>2</sub> Clean	
Fuel Density (g/cm <sup>3</sup> )	9.95	9.95	9.17	9.62
Exposure Gwd/t	0	34	0	0
<u>Atom Densities (atoms/b-cm)</u>				
NAT <sub>80</sub>	0.04372	0.04372	0.04184	0.04215
<sup>16</sup> O	0.04361	0.04361	0.04174	0.04205
<sup>17</sup> O	1.6614-5	1.6614-5	1.5899-5	1.6017-5
<sup>18</sup> O	8.9189-5	8.9189-5	8.5354-5	8.5986-5
<sup>41</sup> Nb	0	7.893-4	0	0
<sup>59</sup> Pr	0	7.893-4	0	0
<sup>90</sup> Th	0	0	0.02025	0
<sup>92</sup> U	0.02186	0.02085	6.724-4	0.01887
<sup>93</sup> Np	0	1.043-5	0	0
<sup>94</sup> Pu	0	2.037-4	0	0.002634
<sup>95</sup> Am	0	5.692-6	0	0
<sup>96</sup> Cm	0	1.131-6	0	0

TABLE II

DATA OF NORTHCLIFFE AND SCHILLING<sup>a</sup> AND ZIEGLER<sup>b</sup> USED IN  
CALCULATING THE ALPHA PARTICLE STOPPING CROSS SECTION OF PLUTONIUM

$E_{\alpha}$ MeV	Stopping Power Ratios and Values from Northcliffe and Schilling				$\epsilon(E)$ Stopping Cross Section	
	$(dE/d\chi)_{Pu}$	$(MeV/mg/cm^2)$		$(dE/d\chi)_{Pu}$	$eV/(10^{15} \text{ atoms/cm}^2)$	
	$(dE/d\chi)_{Al}$	$(dE/d\chi)_{Al}$	$(dE/d\chi)_{U}$	$(dE/d\chi)_{U}$	U(Ziegler)	Pu(Calculated)
0.100	0.150	0.752	0.135	0.837	75.80	63.74
0.320	0.188	1.219	0.243	0.942	139.93	132.48
0.500	0.214	1.317	0.286	0.986	165.64	164.08
0.805	0.235	1.299	0.312	0.978	178.59	175.40
1.281	0.256	1.170	0.307	0.977	166.77	163.72
2.402	0.291	0.904	0.269	0.978	129.15	126.86
4.003	0.322	0.682	0.223	0.982	100.57	99.23
6.404	0.350	0.512	0.183	0.978	78.65	77.29
10.007	0.382	0.379	0.148	0.980	60.67	59.71
16.010	0.418	0.270	0.114	0.991	47.09	46.90
24.016	0.448	0.200	0.090	1.000	37.01	37.18
48.031	0.490	0.118	0.059	0.983	23.64	23.35

<sup>a</sup>Northcliffe and Schilling, Nucl. Data Tables A7, 233 (1970)

<sup>b</sup>J. F. Ziegler, Helium Stopping Powers and Ranges in All Elemental Matter, Vol. 4 of The Stopping and Ranges of Ions In Matter Series (Pergamon Press, New York, 1977).

were fit to each set of tabulated stopping cross-section values, representing the values within 1% at any energy over the range  $0.5 \text{ MeV} \leq E_\alpha \leq 10 \text{ MeV}$ . These functional stopping cross sections are shown in Fig. 1. Coefficients of the polynomial functions are given in Table III. Stopping cross sections of the oxide fuels were formed from these component stopping cross-section functions using the Bragg-Kleeman relationship of Eq. (11) and component densities given in Table I.

Stopping cross-section values of  $\text{UO}_2$ ,  $\text{ThO}_2$ , and  $(\text{U}_{.8}\text{Pu}_{.2})\text{O}_2$  were computed over the range  $2 \text{ MeV} \leq E_\alpha \leq 8 \text{ MeV}$  and compared in Table IV with values of  $\epsilon$  converted from experimentally measured values of  $dE/dx$  reported by Nitzki and Matzke.<sup>7</sup> The measured and calculated values of  $\epsilon$  agree within 9% over this range, with calculated values generally lower than measured values.

### B. ( $\alpha, n$ ) Cross Sections

The cross sections for the  $^{17,18}\text{O}(\alpha, n)$  reactions have been reported over four limited ranges of  $E_\alpha$ , although no single measurement extends over the entire range of our interest. Bair and Willard<sup>14</sup> plotted their measured  $^{18}\text{O}(\alpha, n)^{21}\text{Ne}$  cross-section values over the range  $2.37 \text{ MeV} \leq E_\alpha \leq 5.15 \text{ MeV}$ . Bair and Hass<sup>15</sup> extended the range of these data down to 1.14 MeV and plotted the  $^{17}\text{O}(\alpha, n)^{20}\text{Ne}$  cross section over the range  $1.31 \text{ MeV} \leq E_\alpha \leq 5.31 \text{ MeV}$ . Bair and del Campo<sup>16</sup> later plotted the  $^{\text{NAT}}\text{O}(\alpha, n)$  cross section over the range  $3.1 \text{ MeV} \leq E_\alpha \leq 8 \text{ MeV}$  and, based on their measured  $^{\text{NAT}}\text{O}(\alpha, n)$  neutron production by alpha particles in the range  $4.62 \text{ MeV} \leq E_\alpha \leq 4.8 \text{ MeV}$ , recommended that the  $^{17,18}\text{O}(\alpha, n)$  cross sections reported in Refs. 14 and 15 be increased by 35%.

Differential cross sections  $d\sigma(E)/d\Omega$  for  $^{17,18}\text{O}(\alpha, n)$  reactions were measured at higher energies by Hansen et al.,<sup>17</sup> who fit their measured angular distributions with Legendre polynomial expansions that they integrated to yield total  $\sigma(\alpha, n)$  values. These values were plotted for the range  $4.3 \text{ MeV} \leq E_\alpha \leq 12.3 \text{ MeV}$ , and smooth curves were plotted approximating each set of data.

Except for cross-section values given by Hansen et al.<sup>17</sup> at 9.8, 11.6, and 12.3 MeV, no data were available in other than graphic form--despite the best efforts of Bair,<sup>18</sup> del Campo,<sup>19</sup> and Hansen<sup>20</sup> to resurrect their numerical data. Data taken from the  $^{17,18}\text{O}(\alpha, n)$  cross-section curves of Refs. 14 and 15 for the earlier HEDL work<sup>1</sup> were supplied to us.<sup>21</sup> These data were thinned to 744 values of the  $^{17}\text{O}(\alpha, n)$  cross section and 687 values of the  $^{18}\text{O}(\alpha, n)$  cross section. Fourth-degree polynomial fits were made to data taken from the

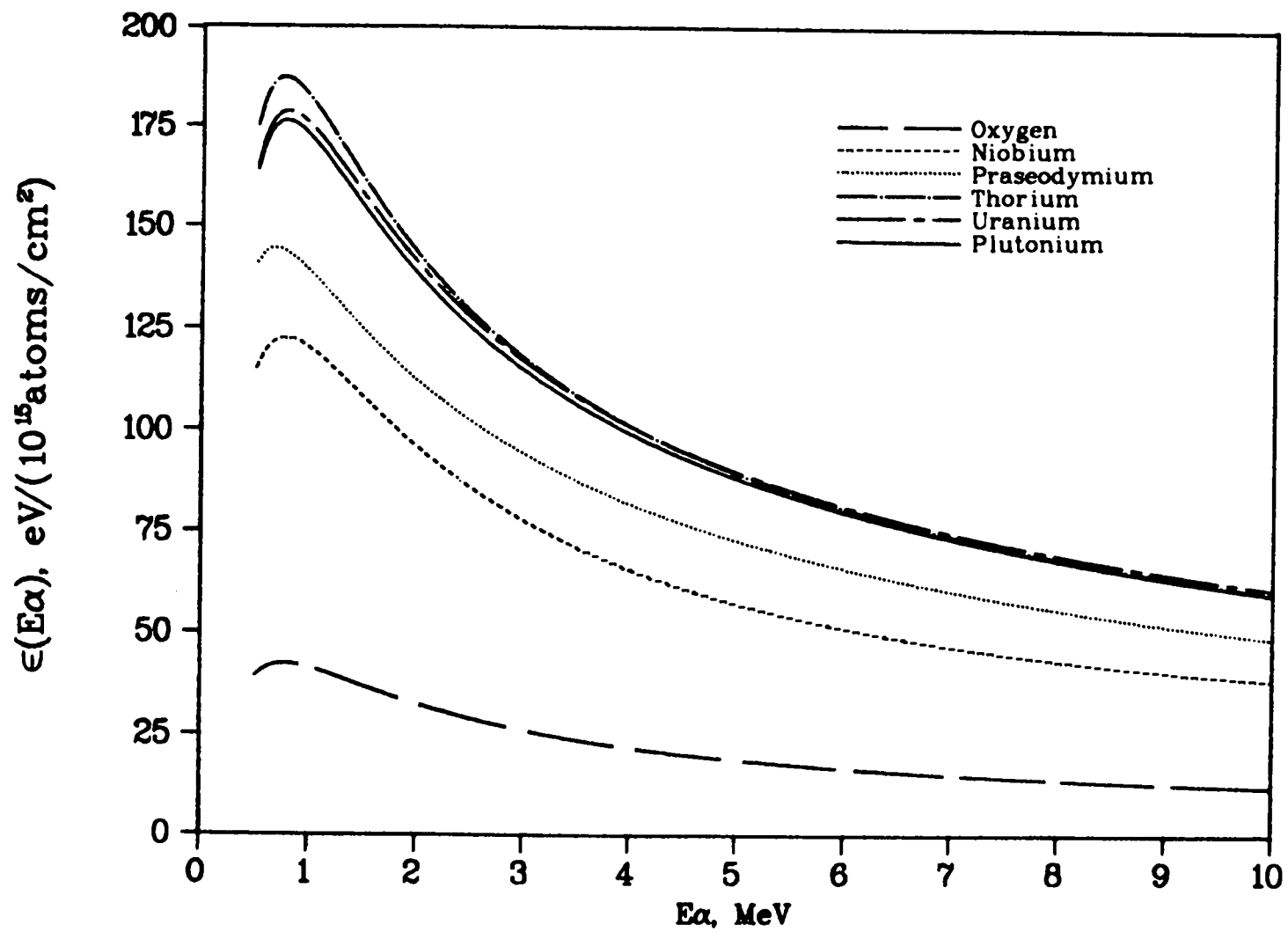


Fig. 1.  
Stopping cross sections  $\epsilon(E_\alpha)$  of O, Nb, Pr, Th, U, and Pu.

TABLE III

COEFFICIENTS OF POLYNOMIAL FITS TO STOPPING CROSS SECTIONS<sup>a</sup>

Element	C <sub>0</sub>	C <sub>1</sub>	C <sub>2</sub>	C <sub>3</sub>	C <sub>4</sub>
O	3.7213	-0.168700	-0.300138	0.0700466	-0.00377296
Ni	4.7872	-0.156294	-0.278932	0.0533399	0.00186590
Pr	4.9321	-0.192312	-0.199561	0.0592391	-0.00940776
Th	5.2027	-0.195369	-0.278809	0.105037	-0.0163945
U	5.1648	-0.161478	-0.279242	0.099232	-0.0146254
Pu	5.1486	-0.171158	-0.272723	0.100975	-0.0160365

<sup>a</sup>  $\ln \epsilon = C_0 + C_1 \ln E + C_2 \ln^2 E + C_3 \ln^3 E + C_4 \ln^4 E$ ,  
 E is alpha-particle energy in MeV,  $0.5 < E \text{ (MeV)} < 10.0$ , and  
 $\epsilon$  is stopping cross section in  $\text{eV}/(10^{15} \text{ atoms/cm}^2)$ .

TABLE IV

COMPARISON OF CALCULATED AND MEASURED  
ALPHA STOPPING CROSS SECTIONS FOR OXIDE FUELS

E MeV	$\epsilon(E)$ for ThO <sub>2</sub>			$\epsilon(E)$ for UO <sub>2</sub>			$\epsilon(E)$ for (U .8Pu .2)O <sub>2</sub>		
	From a	From Table III and Eq. (11)	% Dif	From a	From Table III and Eq. (11)	% Dif	From a	From Table III and Eq. (11)	% Dif
2	68.96	69.40	0.6	71.10	68.73	-3.3	72.17	68.55	-5.0
3	59.38	56.27	-5.2	59.91	55.93	-6.6	60.48	55.84	-7.7
4	52.13	48.67	-6.6	51.76	48.13	-7.0	52.05	48.01	-7.8
5	46.46	42.43	-8.7	45.56	42.53	-6.6	45.69	42.43	-7.1
6	41.91	38.20	-8.9	40.69	38.37	-5.7	40.71	38.27	-6.0
7	38.16	34.87	-8.6	36.76	35.10	-4.5	36.71	35.00	-4.7
8	35.03	32.23	-8.0	33.52	32.50	-3.0	33.42	32.41	-3.0

<sup>a</sup> Nitzki and Matzke, Phys. Rev. B8, 1894 (1973).

$^{17,18}\text{O}(\alpha,n)$  cross-section plot of Ref. 16 and to data taken from the  $^{17,18}\text{O}(\alpha,n)$  cross-section plots of Ref. 17. These five cross-section descriptions are shown in Fig. 2.

The  $^{17,18}\text{O}(\alpha,n)$  cross sections used in the present calculations were composed of the lower energy data of Refs. 14 and 15 increased by 35% as recommended in Ref. 16 and joined with the adjusted higher energy data of Ref. 17. This adjustment, amounting to a 9.2% reduction, was determined by normalizing the integral of the  $^{17,18}\text{O}(\alpha,n)$  cross section formed from the functional fits to  $^{17,18}\text{O}(\alpha,n)$  cross sections of Ref. 17 to the integral of the  $^{17,18}\text{O}(\alpha,n)$  cross section of Ref. 16 over the range  $5.15 \text{ MeV} \leq E_\alpha \leq 8 \text{ MeV}$ . The resulting adjusted cross sections are shown in Fig. 3. The adjusted  $^{17}\text{O}(\alpha,n)$  cross section is given in Table V, and the adjusted  $^{18}\text{O}(\alpha,n)$  cross section is given in Table VI; cross sections are defined there by interpolation points at low energies ( $\leq 5 \text{ MeV}$ ) and by polynomial functions at higher energies.

### C. Alpha-Decay Data

A total of 144 actinide nuclides produced in reactor fuel have been identified,<sup>22</sup> using data of ENDF/B-V and Refs. 23-25. Of these, 89 decay at least partly by alpha decay. Each nuclide has some L different alpha-particle energies with  $1 \leq L \leq 26$  for the data collection used. Alpha-particle energies in the data collection fall in the range  $3.71 \text{ MeV} \leq E_\alpha \leq 8.78 \text{ MeV}$ . TABLE VII lists the alpha-particle energies and intensities for each nuclide.

### D. Spontaneous-Fission Data

Of the 144 actinide nuclides identified, 40 decay at least partly by spontaneous fission. Values of  $\bar{\nu}_p(\text{SF})$ , the major prompt contribution to  $\bar{\nu}(\text{SF})$ , are given by Manero and Konshin<sup>26</sup> for many of these. These values were used in Fig. 4 to estimate values of  $\bar{\nu}_p(\text{SF})$  for nuclides without data.

Branching fractions  $F^{\text{SF}}$ , if not given in a data reference, were constructed from total and SF half-life values  $T_{1/2}(\text{SF})$  using Eq. (19). Values of  $T_{1/2}(\text{SF})$  given as limiting values were used and quoted without qualification. The values of  $\bar{\nu}(\text{SF})$ ,  $F^{\text{SF}}$ , and  $R(\text{SF})$  for each of the 40 nuclides are given in Table VIII.



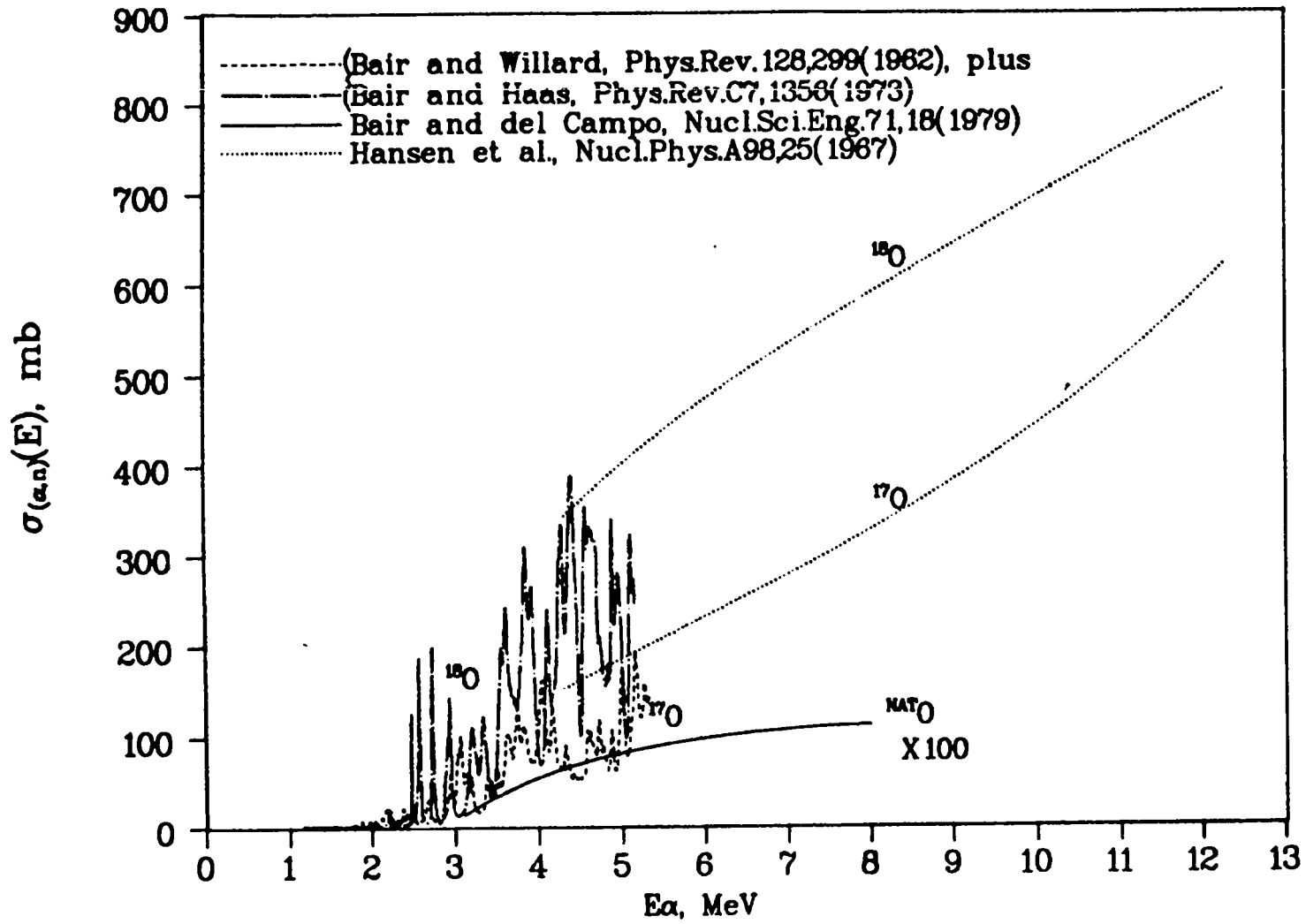


Fig. 2.  
 $^{17}\text{O}$ ,  $^{18}\text{O}$ , and NATO  $(\alpha,n)$  cross-section data.

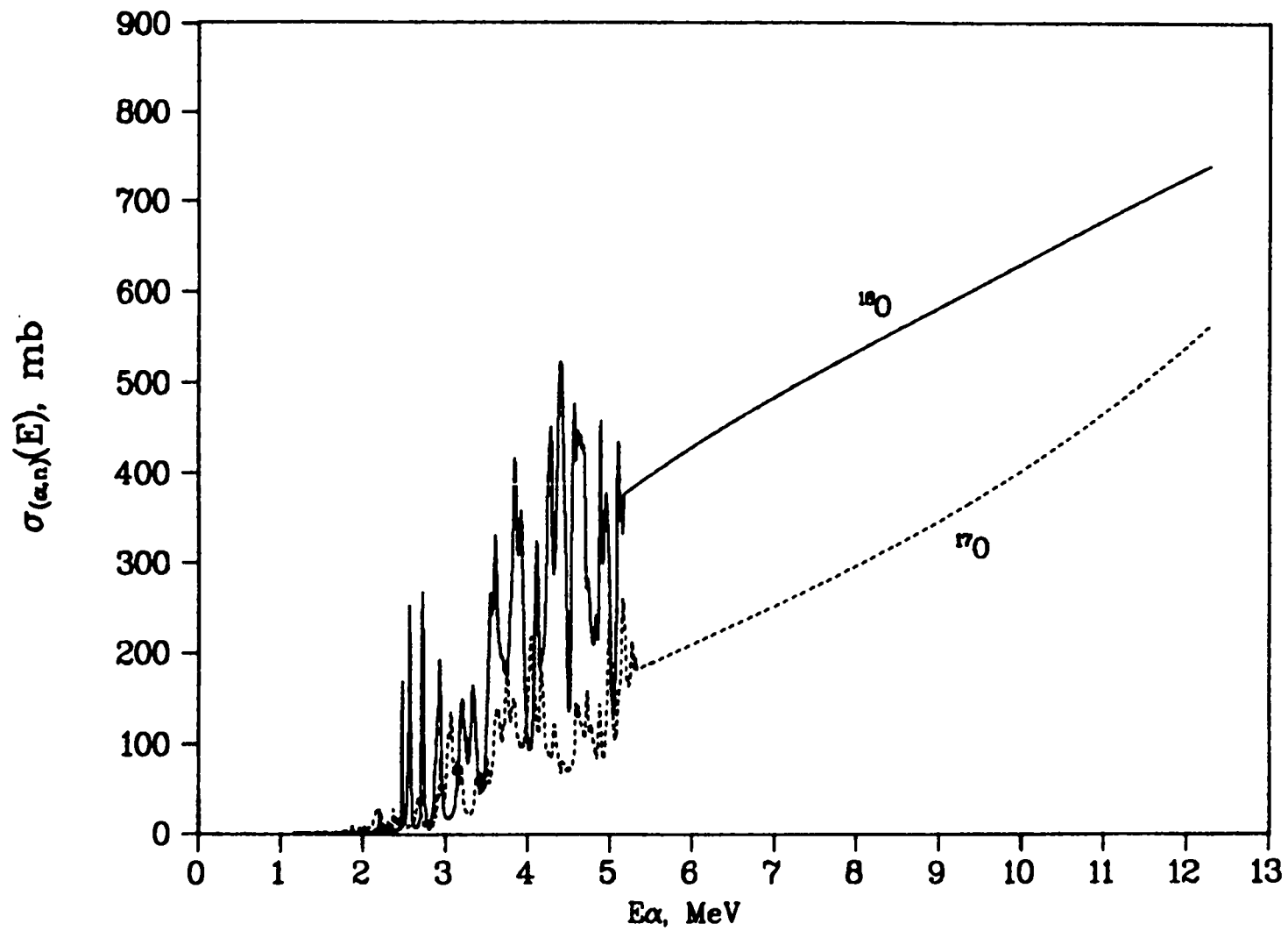


Fig. 3.  
 $^{17}\text{O}$  and  $^{18}\text{O}$  adjusted  $(\alpha, n)$  cross sections.





TABLE VII  
ALPHA DECAY SPECTRA OF ACTINIDE NUCLIDES

82-Pb-210 1 ALPHA E,MEV. DK FRACTION 3.7198 1.70000E-08	83-Bi-214 6 ALPHAS, REF B E,MEV. DK FRACTION 4.9420 5.25210E-07 5.0234 4.41176E-07 5.1824 1.26090E-06 5.2638 1.21849E-05 5.4508 1.13235E-04 5.5121 8.23529E-05	86-Rn-218 2 ALPHAS, REF B E,MEV. DK FRACTION 6.5349 1.30091E-03 7.1331 9.98699E-01	88-Ra-223 12 ALPHAS, REF B E,MEV. DK FRACTION 5.2839 1.00000E-03 5.2885 1.30000E-03 5.3399 1.00000E-03 5.3660 1.10000E-03 5.4347 2.30000E-02 5.5025 1.00000E-02 5.5396 9.10000E-02 5.6073 2.40000E-01 5.7161 5.28900E-01 5.7474 9.10000E-02 5.8577 3.20000E-03 5.8718 8.50000E-03	89-Ac-227 8 ALPHAS, REF B E,MEV. DK FRACTION 4.715 4.30252E-05 4.7701 1.94308E-04 4.7967 1.12421E-04 4.8556 5.13527E-04 4.8733 8.46626E-04 4.9008 1.52670E-05 4.9416 5.55164E-03 4.9538 6.52318E-03	90-Th-230 7 ALPHAS, REF A E,MEV. DK FRACTION 4.2440 5.00746E-04 4.2780 8.01194E-08 4.3720 1.00149E-05 4.4380 3.00448E-04 4.4800 1.20179E-03 4.6210 2.34349E-01 4.6875 7.64136E-01
83-Bi-210 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6479 7.80000E-07 4.6861 5.20000E-07	84-Po-214 2 ALPHAS, REF B E,MEV. DK FRACTION 6.9025 1.00000E-04 7.6873 9.99900E-01	86-Rn-219 4 ALPHAS, REF B E,MEV. DK FRACTION 6.4250 7.49101E-02 6.5310 1.19856E-03 6.5532 1.14862E-01 6.8194 8.09029E-01	88-Ra-224 5 ALPHAS, REF A E,MEV. DK FRACTION 5.0340 3.09345E-05 5.0470 7.19873E-05 5.1610 7.29872E-05 5.4490 4.89914E-02 5.6856 9.50833E-01	89-Ac-227 19 ALPHAS, REF B E,MEV. DK FRACTION 5.5862 1.81892E-03 5.6021 1.71767E-03 5.6136 2.22312E-03 5.6686 2.12207E-02 5.6940 1.51576E-02 5.7017 3.63783E-02 5.7097 8.28618E-02 5.7142 4.95150E-02 5.7575 2.02102E-01 5.7810 2.32417E-03 5.7952 1.13258E-03 5.8076 1.31366E-02 5.8667 2.42522E-02 5.9103 1.71787E-03 5.9166 7.88197E-03 5.9599 3.03153E-02 5.9779 2.32417E-01 6.0089 2.43048E-02 6.0383 2.42522E-01	91-Pa-230 18 ALPHAS, REF B E,MEV. DK FRACTION 4.7653 6.40446E-08 4.7987 9.60672E-04 4.9343 1.28090E-01 4.9726 2.24157E-07 5.0600 1.28090E-07 5.0836 2.24157E-07 5.1190 1.92134E-07 5.1534 1.28090E-07 5.1859 1.60112E-07 5.2173 1.60112E-07 5.2683 1.12078E-06 5.2762 9.60672E-07 5.2880 9.60672E-07 5.3008 5.44381E-06 5.3126 4.16291E-06 5.3263 5.76403E-06 5.3401 4.80336E-06 5.3450 7.36516E-06
84-Po-210 2 ALPHAS, REF B E,MEV. DK FRACTION 4.5168 1.07000E-05 5.3046 9.99989E-01	84-Po-215 3 ALPHAS, REF B E,MEV. DK FRACTION 6.9497 2.20000E-04 6.9559 3.40000E-04 7.3865 9.99440E-01	86-Rn-220 2 ALPHAS, REF A E,MEV. DK FRACTION 5.7490 7.00000E-04 6.2883 9.99300E-01	89-Ac-225 13 ALPHAS, REF B E,MEV. DK FRACTION 5.2871 2.30161E-03 5.4432 1.50105E-03 5.5520 1.00070E-03 5.5809 1.20084E-02 5.6093 1.20084E-02 5.6377 4.35305E-02 5.6824 1.25088E-02 5.7235 3.40238E-02 5.7320 1.01071E-01 5.7921 9.00630E-02 5.7939 1.80126E-01 5.8043 3.00210E-03 5.8299 5.06855E-01	90-Th-227 5 ALPHAS, REF A E,MEV. DK FRACTION 5.1380 5.00050E-04 5.1770 1.80018E-03 5.2110 3.60036E-03 5.3405 2.67027E-01 5.4233 7.27073E-01	91-Pa-231 19 ALPHAS, REF A E,MEV. DK FRACTION 4.5090 3.02822E-05 4.5660 8.07526E-05 4.6000 1.51411E-04 4.6330 1.00941E-03 4.6440 1.00941E-03 4.6820 1.51411E-02 4.7130 1.00941E-02 4.7360 8.47902E-02 4.7960 4.03763E-04 4.8540 1.41317E-02 4.9020 2.01882E-05 4.9340 3.02822E-05 4.9517 2.30145E-01 4.9760 4.03763E-03 4.9860 1.41317E-02 5.0141 2.56390E-01 5.0297 2.01882E-01 5.0320 2.52352E-02 5.0590 1.11055E-01
85-At-219 1 ALPHA, REF B E,MEV. DK FRACTION 6.2733 9.70000E-01	85-At-215 2 ALPHAS, REF B E,MEV. DK FRACTION 7.6285 5.00000E-04 8.0258 9.99500E-01	87-Fr-221 4 ALPHAS, REF B E,MEV. DK FRACTION 6.0752 1.36426E-03 6.1275 1.37335E-01 6.2434 1.11236E-02 6.3411 8.49477E-01	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.9870 7.80172E-04 5.4898 9.99220E-01	90-Th-228 5 ALPHAS, REF A E,MEV. DK FRACTION 5.1380 5.00050E-04 5.1770 1.80018E-03 5.2110 3.60036E-03 5.3405 2.67027E-01 5.4233 7.27073E-01	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01
84-Po-211 3 ALPHAS, REF B E,MEV. DK FRACTION 6.5694 5.40000E-03 6.8914 5.50000E-03 7.4502 9.89100E-01	84-Po-216 2 ALPHAS, REF A E,MEV. DK FRACTION 5.9850 2.10000E-05 6.7785 9.99979E-01	87-Fr-222 1 ALPHA, REF B E,MEV. DK FRACTION 5.7092 1.00000E-03	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6017 5.55000E-02 4.7846 9.44500E-01	90-Th-229 14 ALPHAS, REF B E,MEV. DK FRACTION 4.7626 6.36107E-03 4.7987 1.28231E-02 4.8148 8.48142E-02 4.8390 4.84653E-02 4.8460 5.67447E-01 4.8622 1.81745E-03 4.9017 1.09047E-01 4.9309 1.11066E-03 4.9686 6.46204E-02 4.9795 3.23102E-02 5.0363 2.42326E-03 5.0492 5.25040E-02 5.0534 1.61551E-02 5.0783 1.00969E-04	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01
83-Bi-212 8 ALPHAS, REF B E,MEV. DK FRACTION 5.3024 3.95857E-07 5.3456 3.59870E-06 5.4849 5.39805E-05 5.6069 3.95857E-03 5.6258 5.39805E-04 5.7684 6.00983E-03 6.0511 2.51549E-01 6.0902 9.78847E-02	85-At-217 4 ALPHAS, REF B E,MEV. DK FRACTION 6.4486 4.00100E-04 6.6113 1.00025E-04 6.8134 2.50063E-04 7.0677 9.99250E-01	87-Fr-222 1 ALPHA, REF B E,MEV. DK FRACTION 5.7092 1.00000E-03	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6017 5.55000E-02 4.7846 9.44500E-01	90-Th-229 14 ALPHAS, REF B E,MEV. DK FRACTION 4.7626 6.36107E-03 4.7987 1.28231E-02 4.8148 8.48142E-02 4.8390 4.84653E-02 4.8460 5.67447E-01 4.8622 1.81745E-03 4.9017 1.09047E-01 4.9309 1.11066E-03 4.9686 6.46204E-02 4.9795 3.23102E-02 5.0363 2.42326E-03 5.0492 5.25040E-02 5.0534 1.61551E-02 5.0783 1.00969E-04	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01
84-Po-212 1 ALPHA, REF B E,MEV. DK FRACTION 8.7846 1.00000E+00	86-Rn-217 1 ALPHA, REF B E,MEV. DK FRACTION 7.7426 1.00000E+00	87-Fr-222 1 ALPHA, REF B E,MEV. DK FRACTION 5.7092 1.00000E-03	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6017 5.55000E-02 4.7846 9.44500E-01	90-Th-229 14 ALPHAS, REF B E,MEV. DK FRACTION 4.7626 6.36107E-03 4.7987 1.28231E-02 4.8148 8.48142E-02 4.8390 4.84653E-02 4.8460 5.67447E-01 4.8622 1.81745E-03 4.9017 1.09047E-01 4.9309 1.11066E-03 4.9686 6.46204E-02 4.9795 3.23102E-02 5.0363 2.42326E-03 5.0492 5.25040E-02 5.0534 1.61551E-02 5.0783 1.00969E-04	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01
83-Bi-213 2 ALPHAS, REF B E,MEV. DK FRACTION 5.5508 1.62963E-03 5.8687 2.03704E-02	84-Po-218 2 ALPHAS, REF B E,MEV. DK FRACTION 5.1810 1.10000E-05 6.0027 9.99989E-01	87-Fr-223 1 ALPHA, REF B E,MEV. DK FRACTION 5.3330 4.00000E-05	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6017 5.55000E-02 4.7846 9.44500E-01	90-Th-229 14 ALPHAS, REF B E,MEV. DK FRACTION 4.7626 6.36107E-03 4.7987 1.28231E-02 4.8148 8.48142E-02 4.8390 4.84653E-02 4.8460 5.67447E-01 4.8622 1.81745E-03 4.9017 1.09047E-01 4.9309 1.11066E-03 4.9686 6.46204E-02 4.9795 3.23102E-02 5.0363 2.42326E-03 5.0492 5.25040E-02 5.0534 1.61551E-02 5.0783 1.00969E-04	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01
84-Po-213 2 ALPHAS, REF B E,MEV. DK FRACTION 7.6123 3.00000E-05 8.3757 9.99970E-01	85-At-218 3 ALPHAS, REF B E,MEV. DK FRACTION 6.6625 6.40000E-02 6.7045 9.00000E-01 6.7567 3.60000E-02	87-Fr-223 1 ALPHA, REF B E,MEV. DK FRACTION 5.3330 4.00000E-05	88-Ra-226 2 ALPHAS, REF B E,MEV. DK FRACTION 4.6017 5.55000E-02 4.7846 9.44500E-01	90-Th-229 14 ALPHAS, REF B E,MEV. DK FRACTION 4.7626 6.36107E-03 4.7987 1.28231E-02 4.8148 8.48142E-02 4.8390 4.84653E-02 4.8460 5.67447E-01 4.8622 1.81745E-03 4.9017 1.09047E-01 4.9309 1.11066E-03 4.9686 6.46204E-02 4.9795 3.23102E-02 5.0363 2.42326E-03 5.0492 5.25040E-02 5.0534 1.61551E-02 5.0783 1.00969E-04	92-U-230 4 ALPHAS, REF B E,MEV. DK FRACTION 5.6622 2.30023E-03 5.6661 3.60036E-05 5.8176 3.19032E-01 5.8886 6.75068E-01

TABLE VII (cont.)

92- U-231 1 ALPHA, REF B E, MEV. DK FRACTION 5.4539 5.50000E-05	92- U-235 12 ALPHAS, REF A E, MEV. DK FRACTION 4.1540 9.00000E-03 4.2170 5.70000E-02 4.2270 9.00000E-03 4.2710 4.00000E-03 4.3240 4.70000E-02 4.3640 1.70000E-01 4.3980 5.60000E-01 4.4160 2.10000E-02 4.4390 7.00000E-03 4.5020 1.70000E-02 4.5560 4.50000E-02 4.5980 5.40000E-02	93-NP-237 1 ALPHA, REF A E, MEV. DK FRACTION 4.5140 3.98843E-04 4.5810 3.98843E-03 4.5980 3.99017E-03 4.6391 6.16213E-02 4.6640 3.31040E-02 4.6940 4.78612E-03 4.7120 1.12673E-02 4.7659 7.97687E-02 4.7701 2.49277E-01 4.7882 4.68641E-01 4.8030 2.99133E-02 4.8170 2.49277E-02 4.8660 2.99133E-03 4.8730 2.59248E-02	94-PU-239 21 ALPHAS, REF A E, MEV. DK FRACTION 4.3990 2.50024E-07 4.5100 8.00078E-07 4.6300 7.00068E-06 4.6890 5.00048E-06 4.7360 4.50044E-05 4.7490 6.00058E-06 4.7690 8.00078E-06 4.7950 7.00068E-06 4.8280 2.40023E-05 4.8680 8.00078E-06 4.9110 2.00019E-06 4.9350 3.00029E-05 4.9620 3.00029E-05 4.9870 7.00068E-05 5.0080 8.00078E-05 5.0280 5.00048E-05 5.0540 2.10020E-04 5.0750 3.20031E-04 5.1046 1.15011E-01 5.1429 7.15015E-01 5.1554 1.30371E-01	95-AM-241 21 ALPHAS, REF A E, MEV. DK FRACTION 4.8000 9.00775E-07 4.8340 7.00063E-06 5.0040 1.00086E-06 5.0680 1.40121E-06 5.0840 4.00345E-06 5.0960 4.00345E-06 5.1140 4.00345E-06 5.1560 7.00063E-06 5.1780 3.00258E-06 5.1820 9.00775E-06 5.1940 6.00517E-06 5.2230 1.30112E-05 5.2440 2.40207E-05 5.2790 5.00431E-06 5.3220 1.50129E-04 5.3690 1.33115E-02 5.4170 1.00086E-04 5.4430 1.28110E-01 5.4857 8.52734E-01 5.5130 2.00172E-03 5.5443 3.50302E-03	96-CM-242 8 ALPHAS, REF A E, MEV. DK FRACTION 5.1460 4.99796E-08 5.1840 2.49898E-07 5.5140 1.00086E-06 5.6090 1.99918E-07 5.8170 4.59812E-05 5.9720 3.59653E-04 6.0696 2.59894E-01 6.1129 7.39637E-01	95-AM-243 13 ALPHAS, REF A E, MEV. DK FRACTION 4.6950 1.60296E-05 4.9190 8.51572E-07 4.9300 1.80335E-06 4.9460 3.40629E-06 5.0080 1.60296E-05 5.0320 6.20407E-05 5.0860 4.00740E-05 5.1130 5.80949E-06 5.1810 1.10203E-02 5.2335 1.10203E-01 5.2724 8.75616E-01 5.3210 1.20222E-03 5.3492 1.80333E-03	96-CM-243 28 ALPHAS, REF A E, MEV. DK FRACTION 5.2260 3.86407E-06 5.2600 1.46041E-05 5.3150 9.90940E-06 5.3220 2.97282E-05 5.3310 2.97282E-05 5.5220 1.98188E-05 5.5310 5.94564E-05 5.5360 1.98188E-05 5.5670 6.93658E-05 5.5740 6.93658E-05 5.5810 8.91846E-05 5.5860 1.98188E-04 5.5920 4.90940E-05 5.6080 9.90940E-05 5.6110 3.96376E-04 5.6210 5.94564E-04 5.6380 1.38732E-03 5.6450 2.97282E-04 5.6810 1.98188E-03 5.6850 1.58550E-02 5.7415 1.13956E-01 5.7847 7.25368E-01 5.8750 6.93658E-03 5.9060 9.90940E-04 5.9920 5.64836E-02 6.0090 1.09003E-02 6.0560 4.65742E-02 6.0670 1.46641E-02						
90-TH-232 3 ALPHAS, REF A E, MEV. DK FRACTION 3.8300 1.99601E-03 3.9530 2.29541E-01 4.0120 7.68463E-01	92- U-232 7 ALPHAS, REF A E, MEV. DK FRACTION 4.5090 2.39800E-07 4.9291 2.09825E-06 4.9460 1.69859E-06 4.9973 2.89759E-05 5.1373 2.79767E-03 5.2635 3.11740E-01 5.3203 6.85429E-01	94-PU-237 2 ALPHAS, REF A E, MEV. DK FRACTION 5.3600 2.60700E-05 5.6500 6.93000E-06	92- U-238 3 ALPHAS, REF A E, MEV. DK FRACTION 4.0390 2.29472E-03 4.1490 2.29472E-01 4.1960 7.68233E-01	94-PU-240 5 ALPHAS, REF A E, MEV. DK FRACTION 4.4800 2.10015E-07 4.8510 2.00014E-05 5.0140 9.10064E-04 5.1234 2.65018E-01 5.1683 7.34051E-01	96-CM-241 11 ALPHAS, REF A E, MEV. DK FRACTION 5.6870 2.20044E-05 5.7190 8.00160E-06 5.7850 7.00140E-06 5.8630 1.40028E-05 5.8843 1.18024E-03 5.9140 1.20024E-05 5.9260 1.81036E-04 5.9386 6.89138E-03 5.9780 2.80056E-05 6.0360 1.20024E-05 6.0820 1.50030E-05	94-PU-242 4 ALPHAS, REF A E, MEV. DK FRACTION 4.5985 1.30001E-05 4.7546 9.80007E-04 4.8563 2.24002E-01 4.9006 7.75005E-01	95-AM-242M 7 ALPHAS, REF A E, MEV. DK FRACTION 5.0650 1.20056E-05 5.0860 1.53672E-05 5.1400 2.90537E-04 5.2050 4.27401E-03 5.3130 3.84181E-05 5.3650 7.20339E-05 5.4080 5.76271E-05	94-PU-241 11 ALPHAS, REF A E, MEV. DK FRACTION 4.6920 7.35662E-09 4.7320 7.35662E-09 4.7420 1.71654E-08 4.7830 4.90441E-08 4.7970 2.94265E-07 4.8535 2.96717E-06 4.8965 2.04024E-05 4.9720 3.18787E-07 4.9990 1.00540E-07 5.0420 2.50125E-07 5.0540 8.58272E-08	94-PU-238 11 ALPHAS, REF A E, MEV. DK FRACTION 4.5670 1.10071E-09 4.5880 1.60104E-08 4.6620 3.00195E-08 4.6640 3.00195E-09 4.7030 9.00584E-07 4.7260 1.00065E-07 5.0110 7.00455E-08 5.2080 5.00325E-05 5.3590 1.30084E-03 5.4565 2.87186E-01 5.4992 7.11462E-01	94-PU-235 1 ALPHA, REF B E, MEV. DK FRACTION 5.8556 1.30000E-04	92- U-236 3 ALPHAS, REF A E, MEV. DK FRACTION 4.3310 2.59326E-03 4.4450 2.59326E-01 4.4940 7.38081E-01	94-PU-236 6 ALPHAS, REF A E, MEV. DK FRACTION 5.0880 6.00103E-06 5.2140 2.70046E-06 5.4520 2.00034E-05 5.6150 1.80031E-03 5.7210 3.09053E-01 5.7680 6.89118E-01	92- U-234 5 ALPHAS, REF A E, MEV. DK FRACTION 4.1200 3.29013E-07 4.2740 4.48654E-07 4.6030 2.99102E-03 4.7228 2.74177E-01 4.7730 7.22831E-01

TABLE VII (cont.)

94-PU-244		97-BK-249		98-CF-252		99-ES-254	
2 ALPHAS, REF A		7 ALPHAS, REF A		5 ALPHAS, REF A		8 ALPHAS, REF B	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV.	DK FRACTION
4.5460	1.93757E-01	5.0450	1.45291E-08	5.6160	5.80623E-07	6.2669	2.21953E-03
4.5890	8.04993E-01	5.1150	3.92285E-07	5.8263	1.93541E-05	6.2758	1.61421E-03
96-CM-244		98-CF-249		98-CF-253		99-ES-254M	
8 ALPHAS, REF A		16 ALPHAS, REF A		2 ALPHAS, REF A		7 ALPHAS, REF B	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV.	DK FRACTION
4.9200	8.99767E-07	5.3510	1.99410E-05	5.9210	1.64300E-04	6.2472	7.37056E-05
4.9600	1.99948E-06	5.4310	9.97049E-05	5.9790	2.93570E-03	6.2797	2.78071E-04
5.2150	1.19969E-06	5.5020	4.38701E-04	99-ES-253		6.3045	2.51269E-03
5.3130	3.99896E-07	5.5590	1.09675E-03	28 ALPHAS, REF A		6.3405	6.03046E-05
5.5130	3.49909E-05	5.6230	1.99410E-04	E, MEV. DK FRACTION		6.4361	4.69036E-05
5.6640	2.19943E-04	5.6940	2.99115E-03	5.7300	8.00043E-07	6.4791	1.94315E-04
5.7628	2.35939E-01	5.7597	4.78583E-02	5.9100	2.70015E-07	6.5130	1.34010E-04
5.8050	7.63802E-01	5.7840	2.49262E-03	5.9350	4.00022E-07	100-FM-254	
96-CM-245		98-CF-250		99-ES-253		3 ALPHAS, REF B	
6 ALPHAS, REF A		4 ALPHAS, REF A		28 ALPHAS, REF A		E, MEV. DK FRACTION	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
5.2346	3.20256E-03	5.7367	9.99130E-05	5.9440	1.50008E-06	7.0492	9.00369E-03
5.3038	4.97398E-02	5.8495	1.39587E-02	6.0190	1.80010E-06	7.1468	1.40057E-01
5.3620	9.32546E-01	5.9034	3.19056E-02	6.0370	2.90016E-06	7.1889	8.50349E-01
5.4363	4.00320E-04	5.9462	3.38997E-02	6.0460	4.00022E-06	99-ES-255	
5.4887	8.30665E-03	6.0000	5.98229E-04	6.0840	2.50013E-06	3 ALPHAS, REF B	
5.5292	5.80464E-03	6.0720	3.98819E-03	6.1000	3.40018E-05	E, MEV. DK FRACTION	
96-CM-246		98-CF-251		99-ES-255		DK FRACTION	
2 ALPHAS, REF A		14 ALPHAS, REF A		3 ALPHAS, REF B		DK FRACTION	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
5.3430	2.09945E-01	5.7367	9.99130E-05	6.2170	1.50008E-05	6.2137	2.00000E-03
5.3860	7.89793E-01	5.8495	1.39587E-02	6.2300	1.20006E-06	6.2609	7.84000E-03
96-CM-247		98-CF-251		99-ES-255		6.2996	
7 ALPHAS, REF A		14 ALPHAS, REF A		3 ALPHAS, REF B		7.01600E-02	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
4.8140	4.70000E-02	5.5010	3.04569E-03	6.2500	4.50024E-04	100-FM-255	
4.8680	7.10000E-01	5.5660	1.52284E-02	6.2660	8.00043E-06	6 ALPHAS, REF B	
4.9410	1.60000E-02	5.6030	2.03046E-03	6.3250	4.00022E-06	E, MEV. DK FRACTION	
4.9830	2.00000E-02	5.6320	4.56853E-02	6.3540	8.20044E-05	DK FRACTION	
5.1450	1.20000E-02	5.6480	3.55330E-02	6.4080	1.30007E-04	6.2609	7.84000E-03
5.2100	5.70000E-02	5.6770	3.53299E-01	6.4320	8.10033E-04	6.2996	7.01600E-02
5.2650	1.38000E-01	5.7380	1.01523E-02	6.4800	8.50046E-04	100-FM-255	
96-CM-248		98-CF-254		99-ES-255		6 ALPHAS, REF B	
2 ALPHAS, REF A		2 ALPHAS, REF B		3 ALPHAS, REF B		E, MEV. DK FRACTION	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
5.0340	1.66049E-01	5.6770	3.53299E-01	6.4980	8.50046E-03	6.8069	1.10375E-03
5.0780	7.51351E-01	5.7380	1.01523E-02	6.5400	8.50046E-03	6.8915	6.22115E-03
98-CF-248		98-CF-254		99-ES-255		6.9634	
2 ALPHAS, REF B		2 ALPHAS, REF B		3 ALPHAS, REF B		5.01706E-02	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		6.9829	
6.2241	1.80000E-01	5.7620	3.85787E-02	6.5520	7.10038E-03	7.0225	
6.2663	8.20000E-01	5.7930	2.03046E-02	6.5920	6.60036E-02	4.01365E-03	
98-CF-248		98-CF-254		99-ES-255		6.8069	
2 ALPHAS, REF B		2 ALPHAS, REF B		3 ALPHAS, REF B		1.10375E-03	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		6.22115E-03	
6.2241	1.80000E-01	5.8140	4.26396E-02	6.6240	8.00043E-03	5.01706E-02	
6.2663	8.20000E-01	5.8520	2.78173E-01	6.6327	8.98048E-01	6.9829	
98-CF-248		98-CF-254		99-ES-255		9.37186E-01	
2 ALPHAS, REF B		2 ALPHAS, REF B		3 ALPHAS, REF B		7.0800	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		4.01365E-03	
6.2241	1.80000E-01	5.9430	6.09137E-03	100-FM-256		1 ALPHA, REF B	
6.2663	8.20000E-01	6.0140	1.21827E-01	1 ALPHA, REF B		E, MEV. DK FRACTION	
98-CF-248		98-CF-254		99-ES-255		DK FRACTION	
2 ALPHAS, REF B		2 ALPHAS, REF B		3 ALPHAS, REF B		DK FRACTION	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
6.2241	1.80000E-01	6.0740	2.74112E-02	5.7943	5.27000E-04	6.9152	1.00000E+00
6.2663	8.20000E-01	98-CF-254		99-ES-255		100-FM-257	
98-CF-248		2 ALPHAS, REF B		3 ALPHAS, REF B		5 ALPHAS, REF B	
2 ALPHAS, REF B		E, MEV. DK FRACTION		E, MEV. DK FRACTION		E, MEV. DK FRACTION	
E, MEV.	DK FRACTION	E, MEV.	DK FRACTION	E, MEV. DK FRACTION		DK FRACTION	
6.2241	1.80000E-01	6.3467	3.01811E-03	6.3467		3.01811E-03	
6.2663	8.20000E-01	6.4410	2.01207E-02	6.4410		2.01207E-02	
98-CF-248		6.5199		6.5199		9.35614E-01	
2 ALPHAS, REF B		9.35614E-01		6.6965		3.52113E-02	
E, MEV. DK FRACTION		3.52113E-02		6.7572		6.03622E-03	
E, MEV.	DK FRACTION	6.03622E-03		6.7572		6.03622E-03	

REFERENCE A = ENDFB/B-V  
 REFERENCE B = TABLE OF ISOTOPES, SEVENTH EDITION

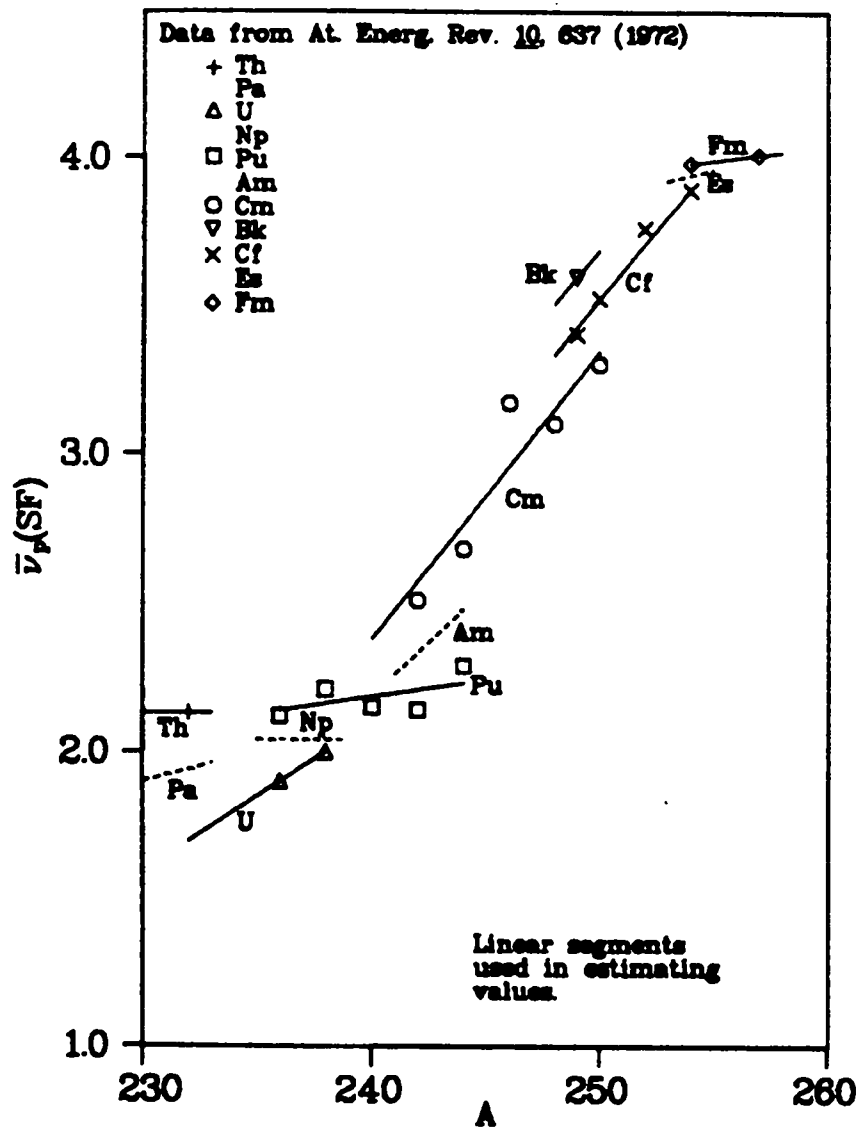


Fig. 4.  
Values of  $\bar{\nu}_p(SF)$ .



TABLE VIII

## SPONTANEOUS-FISSION NEUTRON PRODUCTION BY ACTINIDE DECAY

NUCLIDE	NU-BAR VALUES			SPONTANEOUS FISSION BRANCHING	NEUTRONS PER NUCLIDE DECAY
	PROMPT	DELAYED	TOTAL		
90-TH-230	2.13	.01	2.14	5.330-13 A	1.14 -12
91-PA-231	1.92	.01	1.93	2.980-12 A	5.75 -12
90-TH-232	2.130+.200 B	.01	2.14	1.410-11 A	3.02 -11
92- U-232	1.70	.01	1.71	9.000-13 C	1.54 -12
92- U-233	1.75	.01	1.76	1.300-12 C	2.29 -12
92- U-234	1.80	.01	1.81	1.200-11 C	2.17 -11
92- U-235	1.85	.01	1.86	2.011-09 A	3.74 -09
92- U-236	1.900+.050 B	.01	1.91	1.200-09 C	2.29 -09
94-PU-236	2.120+.130 B	.01	2.13	8.100-10 C	1.73 -09
93-NP-237	2.04	.01	2.05	2.140-12 A	4.39 -12
92- U-238	2.000+.030 B	.01	2.01	5.450-07 C	1.095-06
94-PU-238	2.210+.130 B	.01	2.22	1.840-09 C	4.08 -09
94-PU-239	2.15	.01	2.16	4.400-12 C	9.37 -12
94-PU-240	2.151+.006 B	.01	2.16	5.000-08 C	1.08 -07
96-CM-240	2.38	.01	2.39	3.860-08 A	9.23 -08
95-AM-241	2.26	.01	2.27	4.100-12 C	9.31 -12
94-PU-242	2.141+.190 B	.01	2.15	5.500-06 C	1.18 -05
95-AM-242M	2.33	.01	2.34	1.600-10 C	3.74 -10
96-CM-242	2.510+.060 B	.01	2.52	6.800-08 C	1.71 -07
95-AM-243	2.41	.01	2.42	2.200-10 C	5.32 -10
94-PU-244	2.290+.190 B	.01	2.30	1.250-03 C	2.88 -03
96-CM-244	2.681+.011 B	.01	2.69	1.347-06 C	3.62 -06
96-CM-246	3.170+.220 B	.01	3.18	2.614-04 C	8.31 -04
96-CM-248	3.100+.090 B	.01	3.11	8.260-02 C	2.569-01
98-CF-248	3.33	.01	3.34	2.850-05 A	9.52 -05
97-Bk-249	3.590+.160 B	.01	3.60	4.600-10 C	1.66 -09
98-CF-249	3.400+.400 B	.01	3.41	5.020-09 A	1.71 -08
96-CM-250	3.300+.080 B	.01	3.31	7.000-01 D	2.32 +00
98-CF-250	3.520+.090 B	.01	3.53	3.092-02 C	2.72 -03
98-CF-252	3.756+.012 B	.009 B	3.765+.010 B	3.092-02 C	1.164-01
99-ES-253	3.92	.01	3.93	8.700-08 C	3.42 -07
98-CF-254	3.890+.050 B	.01	3.890+.050 E	9.969-01 A	3.88 +00
99-ES-254	3.94	.01	3.95	3.020-08 A	1.19 -07
99-ES-254M	3.94	.01	3.95	4.500-08 A	1.78 -07
100-FM-254	3.980+.140 B	.01	3.96 +.14 F	5.900-04 A	2.34 -03
99-ES-255	3.96	.01	3.97	4.000-05 A	1.59 -04
100-FM-255	3.99	.01	3.73 +.18 F	2.290-07 A	8.54 -07
100-FM-256	4.00	.01	4.01	9.190-01 A	3.69 +00
100-FM-257	4.010+.130 B	.01	3.85 +.05 G	2.100-03 A	8.09 -03
100-FM-258	4.02	.01	4.03	1.000+00 A	4.03 +00

## DATA REFERENCES USED

A=TABLE OF ISOTOPES, SEVENTH EDITION

B=MANEPO AND KONSHIN, ATOMIC ENERGY REV. 10,637-756(1972)

C=ENDF/B-V

D=A. TOBIAS, U.K., PRIVATE COMMUNICATION

E=C. J. ORTH, NUCL. SCI. ENG. 43, 54 (1971)

F=Y. A. LAZAREV, ATOMIC ENERGY REV. 15, 75 (1977)

G=D. C. HOFFMAN ET AL., PHYS. REV. C21, 637 (1980)

## ADDITIONAL REFERENCES SURVEYED

J. U. BOLDIEMAN, IN NEUTRON STD. REF. DATA, I. A. E. A. VIENNA (1974)

J. P. BALAGNA ET AL., PHYS. REV. LETT. 26, 145 (1971)

PROMPT NU-BAR VALUES GIVEN WITHOUT REFERENCE HAVE BEEN ESTIMATED FROM THE VALUES OF REFERENCE B. DELAYED NU-BAR VALUES GIVEN WITHOUT REFERENCE HAVE BEEN ARBITRARILY ASSUMED.

#### IV. CALCULATION OF THE THICK-TARGET NEUTRON-PRODUCTION FUNCTION $P_i(E_\alpha)$

The neutron-production function  $P_i(E_\alpha)$  defined by Eqs. (6) and (9) gives the contribution from reactions with nuclide  $i$  to the probability of neutron production by a decay alpha particle of energy  $E_\alpha$  emitted within the material. The POFEAL code calculates values of  $P_i$  OF E-ALPHA using the algorithm

$$P(J) = 1.E + 6* \frac{N_i}{N} \sum_{j=2}^J \frac{[\sigma_i(j-1) + \sigma_i(j)]/2}{[\epsilon(j-1) + \epsilon(j)]/2} [E(j) - E(j-1)] \quad , \quad (25)$$

where

$N_i$  is the atom density of nuclide  $i$  (atoms/cm<sup>3</sup>),

$N$  is the total atom density (atoms/cm<sup>3</sup>),

$E_j$  is the  $j$ th regular energy point at or above the cross-section threshold (MeV),

$\sigma_i(j)$  is the value of the ( $\alpha, n$ ) cross section of nuclide  $i$  at  $E_j$  (mb),

$\epsilon(j)$  is the value of the stopping cross section (eV/10<sup>15</sup> atoms/cm<sup>2</sup>),

and the leading quantity of  $1 \times 10^6$  is required because of the units of  $\sigma$ ,  $\epsilon$ , and  $E$ .

The <sup>17</sup>O and <sup>18</sup>O contributions to the ( $\alpha, n$ ) neutron-production rate are given in Tables IX-XII for each of the four fuel compositions given in Table I. Values for the four compositions at any energy differ by less than 4%. The <sup>17</sup>O and <sup>18</sup>O contributions to ( $\alpha, n$ ) neutron production in spent UO<sub>2</sub> fuel are shown in Fig. 5.

#### V. RESULTS

The half-lives, average decay energies, and spent UO<sub>2</sub> fuel neutron-production values  $R_k(\alpha, n)$ ,  $R_k(\text{SF})$ , and  $R_k$  for each of the actinide nuclides  $k$  are given in Table XIII. Values of  $R_k(\text{SF})$  are repeated from Table VIII. Values of  $R_k(\alpha, n)$  were obtained using the alpha spectra data of Table VII and  $P(E_\alpha)$  values given in Table XI for <sup>17,18</sup>O( $\alpha, n$ ) in spent UO<sub>2</sub> fuel.

TABLE IX

17, 18 O(alpha,n) NEUTRON PRODUCTION IN CLEAN ThO2 FUEL BY ALPHA PARTICLES BELOW 10 MeV

Table with multiple columns: E, MEV, neutrons-per-alpha, and total values for various alpha energy ranges (0-17, 0-18, total) across different alpha energies from 0.000 to 9.999 MeV.







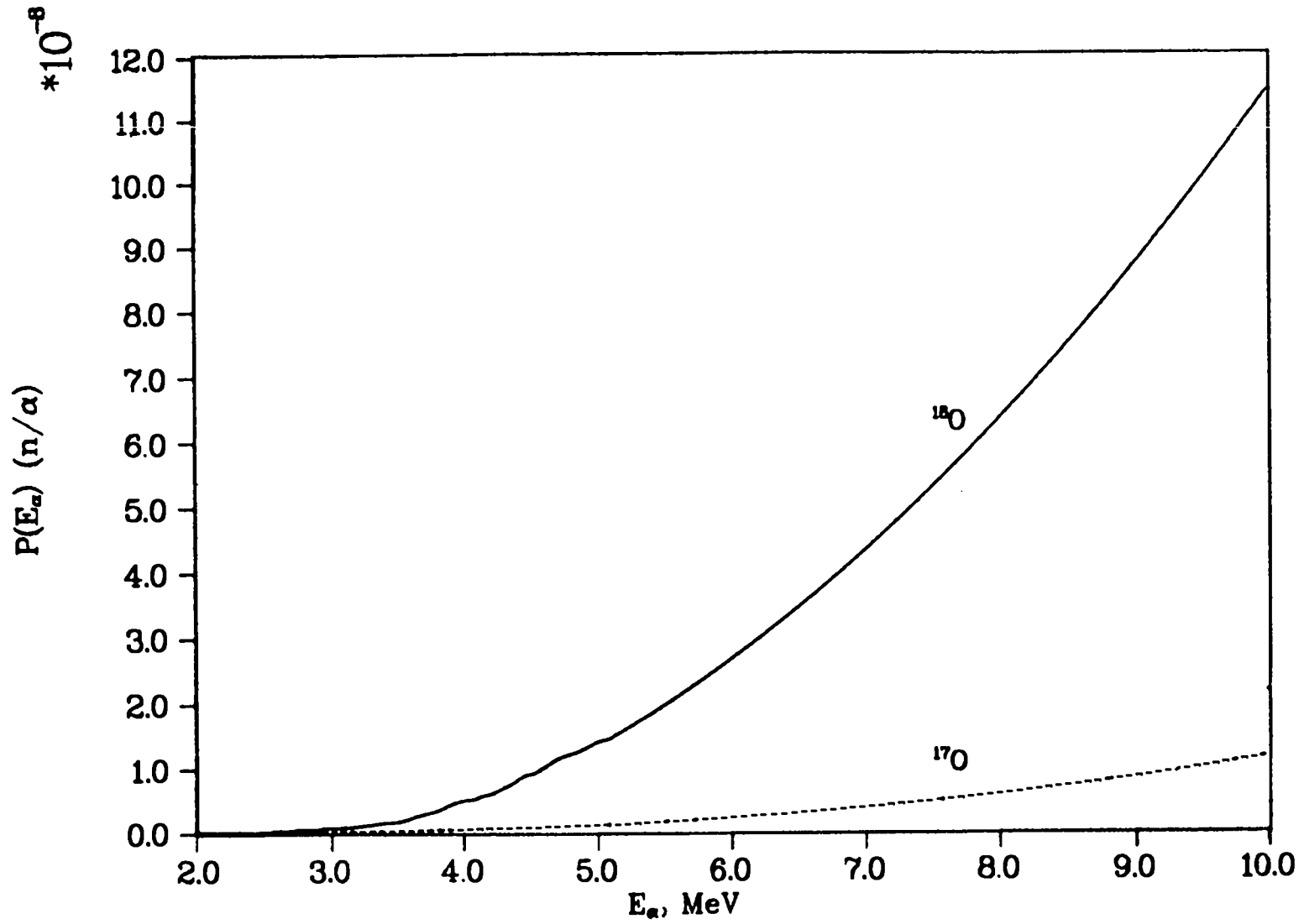


Fig. 5.  
 $^{17,18}\text{O}(\alpha, n)$  neutron production by decay alphas in LWR irradiated  $\text{UO}_2$  fuel.

TABLE XIII

NEUTRON PRODUCTION FROM ACTINIDE DECAY IN UO<sub>2</sub> FUEL

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE- CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆		
				ALPHA IN UO <sub>2</sub>	N FISSION	SPONT. TOTAL
80-HG-206	4.89000+2	0.5274	A	0.	0.	0.
81-TL-206	2.50980+2	0.5402	A	0.	0.	0.
82-PB-206	STABLE	0.	-	0.	0.	0.
81-TL-207	2.87400+2	0.5194	A	0.	0.	0.
82-PB-207	STABLE	0.	-	0.	0.	0.
81-TL-208	1.84200+2	3.9702	B	0.	0.	0.
82-PB-208	STABLE	0.	-	0.	0.	0.
81-TL-209	1.32000+2	2.8315	A	0.	0.	0.
82-PB-209	1.17108+4	0.2234	A	0.	0.	0.
83-BI-209	6.3115+25	0.	A	0.	0.	0.
81-TL-210	7.80000+1	4.2765	A	0.	0.	7.00 -05
82-PB-210	7.02472+8	0.0441	A	5.68 -17	0.	5.68 -17
83-BI-210	4.33123+5	0.3899	A	1.56 -14	0.	1.16 -14
84-PO-210	1.19557+7	5.4076	A	1.87 -08	0.	1.87 -08
82-PB-211	2.16600+3	0.5353	A	0.	0.	0.
83-BI-211	1.29000+2	6.7881	A	3.88 -08	0.	3.88 -08
84-PO-211	0.5160000	7.5942	A	5.64 -08	0.	5.64 -08
82-PB-212	3.83040+4	0.3180	B	0.	0.	0.
83-BI-212	3.63600+3	2.9030	A	1.076-08	0.	1.076-08
84-PO-212	2.96000-7	8.9536	A	8.94 -08	0.	8.94 -08
83-BI-213	2.73540+3	0.7172	A	5.85 -10	0.	5.85 -10
84-PO-213	4.20000-6	8.5360	A	7.86 -08	0.	7.86 -08
82-PB-214	1.60800+3	0.5389	A	0.	0.	0.
83-BI-214	1.18200+3	2.1923	A	4.39 -12	0.	4.39 -12
84-PO-214	1.63700-4	7.8337	A	6.19 -08	0.	6.19 -08
83-BI-215	4.44000+2	0.8445	A	0.	0.	0.
84-PO-215	1.77800-3	7.5265	A	5.52 -08	0.	5.52 -08
85-AT-215	1.00000-4	8.1780	A	6.98 -08	0.	6.98 -08
84-PO-216	0.1500000	6.9064	B	4.28 -08	0.	4.28 -08
85-AT-217	0.0323000	7.2004	A	4.85 -08	0.	4.85 -08
86-RN-217	5.40000-4	7.8880	A	6.32 -08	0.	6.32 -08
84-PO-218	1.83000+2	6.1149	A	2.909-08	0.	2.909-08
85-AT-218	1.7500000	6.8830	A	4.14 -08	0.	4.14 -08
86-RN-218	0.0350000	7.2664	A	4.99 -08	0.	4.99 -08
85-AT-219	5.40000+1	6.2165	A	3.26 -08	0.	3.26 -08
86-RN-219	3.9600000	6.9463	A	4.25 -08	0.	4.25 -08
86-RN-220	5.56000+1	6.4048	B	3.39 -08	0.	3.39 -08
87-FR-221	2.88000+2	6.4580	A	3.45 -08	0.	3.45 -08
86-RN-222	3.30351+5	5.5905	A	2.129-08	0.	2.129-08
87-FR-222	8.64000+2	0.7450	A	2.45 -11	0.	2.45 -11
88-RA-222	3.80000+1	6.6760	A	3.846-08	0.	3.846-08
87-FR-223	1.30800+3	0.4559	A	7.65 -13	0.	7.65 -13
88-RA-223	9.87949+5	-----	A	2.39 -08	0.	2.39 -08
88-RA-224	3.16224+5	5.7903	B	2.40 -08	0.	2.40 -08
88-RA-225	1.27872+6	0.1433	A	0.	0.	0.
89-AC-225	8.64000+5	5.8354	A	2.57 -08	0.	2.57 -08
88-RA-226	5.0461+10	4.8708	A	1.304-08	0.	1.304-08
89-AC-226	1.04400+5	0.4099	A	1.24 -12	0.	1.24 -12



TABLE XIII (cont.)

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE- CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆			TOTAL
				ALPHA	N	SPONT. FISSION	
=====	=====	=====	=====	=====	=====	=====	=====
90-TH-226	1.85400+3	6.4517	A	3.42	-08	0.	3.42 -08
89-AC-227	6.87097+8	0.0878	A	2.01	-10	0.	2.01 -10
90-TH-227	1.61720+6	6.1466	A	2.72	-08	0.	2.72 -08
88-PA-228	1.82087+8	0.0146	A	0.	0.	0.	0.
89-AC-228	2.20680+4	1.3696	A	0.	0.	0.	0.
90-TH-228	6.03725+7	5.5176	B	2.004	-08	0.	2.004-08
90-TH-229	2.3163+11	5.1686	A	1.391	-08	0.	1.391-08
90-TH-230	2.4299+12	4.7609	B	1.207	-08	1.14 -12	1.21 -08
91-PA-230	1.52928+6	0.6577	A	6.03	-13	0.	6.03 -13
92- U-230	1.79712+6	5.9928	A	2.69	-08	0.	2.69 -08
90-TH-231	9.18720+4	0.1537	B	0.	0.	0.	0.
91-PA-231	1.0338+12	5.0601	B	1.478	-08	5.75 -12	1.48 -08
92- U-231	3.62880+5	0.1017	A	1.14	-12	0.	1.14 -12
90-TH-232	4.4337+17	4.0862	B	5.52	-09	3.02 -11	5.55 -09
91-PA-232	1.13184+5	1.098	B	0.	0.	0.	0.
92- U-232	2.26263+9	5.4145	B	1.871	-08	1.54 -12	1.87 -08
90-TH-233	1.33800+3	0.4422	B	0.	0.	0.	0.
91-PA-233	2.33280+6	0.4080	B	0.	0.	0.	0.
92- U-233	5.0232+12	4.8978	B	1.336	-08	2.29 -12	1.34 -08
90-TH-234	2.08233+6	0.1473	A	0.	0.	0.	0.
91-PA-234	2.43000+4	2.2453	A	0.	0.	0.	0.
91-PA-234M	7.05000+1	0.8141	A	0.	0.	0.	0.
92- U-234	7.7188+12	4.8685	B	1.299	-08	2.17 -11	1.301-08
90-TH-235	4.14000+2	-----	A	0.	0.	0.	0.
91-PA-235	1.45200+3	-----	A	0.	0.	0.	0.
92- U-235	2.2210+16	4.6651	B	8.89	-09	3.74 -09	1.26 -08
92- U-235M	1.48080+3	0.0001	A	0.	0.	0.	0.
93-NP-235	3.42230+7	0.0810	A	2.44	-13	0.	2.44 -13
94-PU-235	1.53600+3	5.8675	A	3.48	-12	0.	3.48 -12
92- U-236	7.3890+14	4.5809	B	9.89	-09	2.29 -09	1.218-08
93-NP-236	3.6290+12	0.3390	B	0.	0.	0.	0.
93-NP-236M	8.10000+4	0.1353	B	0.	0.	0.	0.
94-PU-236	8.99688+7	5.8634	B	2.517	-08	1.73 -09	2.69 -08
92- U-237	5.83200+5	0.3103	B	0.	0.	0.	0.
93-NP-237	6.7532+13	4.9470	B	1.303	-08	4.39 -12	1.303-08
94-PU-237	3.94243+6	0.0628	B	6.72	-13	0.	6.72 -13
92- U-238	1.4100+17	4.2755	B	6.64	-09	1.095-06	1.102-06
93-NP-238	1.82908+5	0.7916	B	0.	0.	0.	0.
94-PU-238	2.76912+9	5.5871	B	2.124	-08	4.08 -09	2.532-08
92- U-239	1.41000+3	0.4650	B	0.	0.	0.	0.
93-NP-239	2.03385+5	0.4180	B	0.	0.	0.	0.
94-PU-239	7.6084+11	5.2396	B	1.664	-08	9.37 -12	1.665-08
92- U-240	5.07600+4	0.1755	A	0.	0.	0.	0.
93-NP-240	4.02000+3	1.5755	A	0.	0.	0.	0.
93-NP-240M	4.50000+2	1.0407	A	0.	0.	0.	0.
94-PU-240	2.0670+11	5.3274	B	1.676	-08	1.08 -07	1.25 -07
95-AM-240	1.82880+5	1.0920	B	3.74	-14	0.	3.74 -14
96-CM-240	2.31552+6	6.3844	A	3.37	-08	9.23 -08	1.26 -07
94-PU-241	4.63886+8	0.0054	B	3.39	-13	0.	3.39 -13
95-AM-241	1.3639+10	5.6131	B	2.115	-08	9.31 -12	2.116-08
96-CM-241	2.83392+6	1.1100	B	2.79	-10	0.	2.79 -10
94-PU-242	1.1875+13	4.9812	B	1.406	-08	1.18 -05	1.18 -05
95-AM-242	5.76360+4	0.1944	B	0.	0.	0.	0.
95-AM-242M	4.79665+9	0.0631	B	9.22	-11	3.74 -10	4.56 -10

TABLE XIII (cont.)

NUCLIDE	HALF-LIFE (SECONDS)	DECAY ENERGY (MEV)	DE- CAY REF	◆◆◆◆NEUTRONS PER DECAY◆◆◆◆		
				ALPHA, IN UO <sub>2</sub>	N FISSION	SPONT. FISSION
96-CM-242	1.40745+7	6.2169	B	3.07 -08	1.714-07	2.02 -07
94-PU-243	1.78452+4	0.1957	B	0.	0.	0.
95-AM-243	2.3289+11	5.4224	B	1.82 -08	5.32 -10	1.87 -08
96-CM-243	8.99372+8	6.1598	B	2.62 -08	0.	2.62 -08
94-PU-244	2.5877+15	4.6510	B	1.083-08	2.875-03	2.88 -03
95-AM-244	3.63600+4	1.1177	B	0.	0.	0.
95-AM-244M	1.56000+3	0.5088	B	0.	0.	0.
96-CM-244	5.71495+8	5.9010	B	2.582-08	3.623-06	3.65 -06
94-PU-245	3.78280+4	0.8103	A	0.	0.	0.
95-AM-245	7.38000+3	0.3199	A	0.	0.	0.
96-CM-245	2.6744+11	5.5881	B	1.948-08	0.	1.95 -08
94-PU-246	9.37440+5	0.2514	A	0.	0.	0.
95-AM-246M	1.50000+3	1.4433	A	0.	0.	0.
96-CM-246	1.4926+11	5.4714	B	1.971-08	8.313-04	8.31 -04
96-CM-247	4.9229+14	5.3522	B	1.466-08	0.	1.47 -08
96-CM-248	1.0720+13	4.7270	B	1.441-08	2.569-01	2.57 -01
97-BK-248	2.84018+8	-----	A	-----	-----	-----
97-BK-248M	8.46000+4	0.1684	A	0.	0.	0.
98-CF-248	2.88144+7	6.3613	A	3.336-08	9.519-05	9.52 -05
96-CM-249	3.84900+3	0.2932	B	0.	0.	0.
97-BK-249	2.76480+7	0.0331	B	2.906-13	1.656-09	1.66 -09
98-CF-249	1.1064+10	6.2903	B	2.646-08	1.712-08	4.36 -08
96-CM-250	3.5660+11	-----	C	-----	2.32 +00	2.32 +00
97-BK-250	1.15812+4	1.1829	B	0.	0.	0.
98-CF-250	4.12764+8	6.1227	B	2.941-08	2.718-03	2.72 -03
96-CM-251	1.00800+3	0.5925	A	0.	0.	0.
97-BK-251	3.33600+3	0.4988	A	0.	0.	0.
98-CF-251	2.8338+10	6.0260	B	2.532-08	0.	2.53 -08
98-CF-252	8.32471+7	6.0317	B	2.996-08	1.164-01	1.164-01
98-CF-253	1.53878+6	0.0980	B	8.89 -11	0.	8.89 -11
99-ES-253	1.76860+6	6.7367	B	3.995-08	3.419-07	3.82 -07
98-CF-254	5.22720+6	0.0184	A	8.167-11	3.88 +00	3.88 +00
99-ES-254	2.38205+7	6.6172	A	3.627-08	1.193-07	1.56 -07
99-ES-254M	1.41480+5	0.7351	A	1.138-10	1.778-07	1.78 -07
100-FM-254	1.16640+4	7.2996	A	5.08 -08	2.34 -03	2.34 -03
98-CF-255	6.84000+3	-----	A	0.	0.	0.
99-ES-255	3.30912+6	0.5956	A	2.72 -09	1.59 -04	1.59 -04
100-FM-255	7.22520+4	7.2407	A	4.75 -08	8.54 -07	9.02 -07
99-ES-256	1.32000+3	0.6169	A	0.	0.	0.
100-FM-256	9.45720+3	7.0250	A	4.55 -08	3.69 +00	3.69 +00
100-FM-257	8.68320+6	6.8640	A	3.81 -08	8.09 -03	8.09 -03
100-FM-258	3.80000-4	-----	A	0.	4.03 +00	4.03 +00

## DECAY DATA REFERENCES

A=TABLE OF ISOTOPES

B=ENDF/B-V

C=A. TOBIAS, U.K., PRIVATE COMMUNICATION

## ADDITIONAL NOTES

MISSING DATA NOTED AS -----

81-TL-210, NEUTRONS FROM DELAYED NEUTRON  
EMISSION FROM 82-PB-210 LEVELS  
PRODUCED IN BETA DECAY.92- U-235, SPONTANEOUS FISSION BRANCHING  
IN ENDF/B-V IS ZERO BY OMISSION.  
S.F. BRANCHING(2.011-9) TAKEN  
FROM REFERENCE A.

97-BK-248 DECAY CHARACTERISTICS UNKNOWN.

These values of  $R_k$  may be used with detailed calculated activity inventory to determine total neutron production within oxide fuel, using Eq. (22).

#### ACKNOWLEDGMENTS

The authors gratefully acknowledge the assistance of P. A. Ombrellaro of Hanford Engineering Development Laboratory in providing information and helpful discussions about the methodology and data used; he, with D. L. Johnson and R. E. Schenter, provided low-energy  $^{17,18}\text{O}(\alpha,n)$  cross-section values used in earlier calculations. We also appreciate conversations with J. K. Bair and J. Gomez del Campo of Oak Ridge National Laboratory and L. F. Hansen of Lawrence Livermore National Laboratory concerning their  $\text{O}(\alpha,n)$  cross-section measurements and related observations.

At Los Alamos, M. E. Battat, R. J. LaBauve, and T. R. England worked on the accumulation of alpha-decay and spontaneous-fission decay data for the actinide nuclides. R. W. Hardie posed the problem and furnished fast-reactor fuel information. D. M. McClellan painstakingly took cross-section values from graphical form, and N. L. Whittemore prepared tables and figures. We sincerely appreciate their contributions.

#### REFERENCES

1. P. A. Ombrellaro and D. L. Johnson, "Subcritical Reactivity Monitoring: Neutron Yields from Spontaneous  $(\alpha,n)$  Reactions in FFTF Fuel," Hanford Engineering Development Laboratory report HEDL TME 78-39 (June 1978). Information in this document was supplemented and updated by personal communications with the authors.
2. W. B. Wilson, T. R. England, R. J. LaBauve, M. E. Battat, D. E. Wessol, and R. T. Perry, "Status of CINDER and ENDF/B-V Based Libraries for Transmutation Calculations," Proc. Int. Conf. on Nuclear Waste Transmutation, Austin, Texas, July 22-24, 1980.
3. R. T. Perry and W. B. Wilson, "The  $(\alpha,n)$  Neutron Production by Alpha Particles in  $\text{PuO}_2$ ,  $\text{UO}_2$ , and  $\text{ThO}_2$  Fuels," in "Applied Nuclear Data Research and Development, April 1-June 30, 1980," Los Alamos Scientific Laboratory report LA-8524-PR (1980), p. 20.
4. R. T. Perry and W. B. Wilson, "Neutron Production from  $(\alpha,n)$  Reactions in  $\text{PuO}_2$ ,  $\text{UO}_2$ , and  $\text{ThO}_2$  Fuels," Trans. Am. Nucl. Soc. 35, 549 (1980).
5. W. B. Wilson, R. T. Perry, T. R. England, R. J. LaBauve, M. E. Battat, and N. L. Whittemore, "Neutron Production from Actinide Decay in Oxide Fuels," in "Applied Nuclear Data Research and Development, July 1-September 30, 1980," Los Alamos Scientific Laboratory report LA-8630-PR (1980), p. 23.

6. H. J. Hirsch and Hj. Matzke, "Stopping Power and Range of  $\alpha$ -Particles in (U,Pu)C and UC and Application to Self-Diffusion Measurements Using Alpha Spectroscopy," J. of Nucl. Mater. 45, 29 (1972).
7. V. Nitzki and Hj. Matzke, "Stopping Power of 1-9 MeV He<sup>++</sup> Ions in UO<sub>2</sub>, (U,Pu)O<sub>2</sub>, and ThO<sub>2</sub>," Phys. Rev. B8, 1894 (1973).
8. L. C. Northcliffe and R. F. Schilling, "Range and Stopping Power Tables for Heavy Ions," Nucl. Data Tables A7, 233 (1970).
9. J. F. Ziegler and W. K. Chu, "Stopping Cross Sections and Backscattering Factors for <sup>4</sup>He Ions in Matter," Atomic Data and Nucl. Data Tables 13, 463 (1974).
10. W. H. Bragg and R. Kleeman, "On the Alpha Particles of Radium and Their Loss of Range in Passing Through Various Atoms and Molecules," Phil. Mag. 10, 318 (1905).
11. H. H. Anderson, Bibliography and Index of Experimental Range and Stopping Power Data, Vol. 2 of The Stopping and Ranges of Ions in Matter Series (Pergamon Press, New York, 1977).
12. H. Bichsel, "A Critical Review of Experimental Stopping Power and Range Data," in "Studies in Penetration of Charged Particles in Matter," National Academy of Sciences - National Research Council publication 1133 (1964).
13. J. F. Ziegler, Helium Stopping Powers and Ranges in All Elemental Matter, Vol. 4 of The Stopping and Ranges of Ions in Matter Series (Pergamon Press, New York, 1977).
14. J. K. Bair and H. B. Willard, "Level Structure in Ne<sup>22</sup> and Si<sup>30</sup> from the Reactions O<sup>18</sup>( $\alpha$ ,n)Ne<sup>21</sup> and Mg<sup>26</sup>( $\alpha$ ,n)Si<sup>29</sup>," Phys. Rev. 128, 299 (1962).
15. J. K. Bair and F. X. Haas, "Total Neutron Yield from the Reactions <sup>13</sup>C( $\alpha$ ,n)<sup>16</sup>O and <sup>17,18</sup>O( $\alpha$ ,n)<sup>20,21</sup>Ne," Phys. Rev. C7, 1356 (1973).
16. J. K. Bair and J. Gomez del Campo, "Neutron Yields from Alpha-Particle Bombardment," Nucl. Sci. Eng. 71, 18 (1979).
17. L. F. Hansen, J. D. Anderson, J. W. McClure, B. A. Pohl, M. L. Stelts, J. J. Wesolowski, and C. Wong, "The ( $\alpha$ ,n) Cross Sections on <sup>17</sup>O and <sup>18</sup>O Between 5 and 12.5 MeV," Nucl. Phys. A98, 25 (1967).
18. J. K. Bair (retired), Oak Ridge National Laboratory, personal communication (June 1980).
19. J. Gomez del Campo, Oak Ridge National Laboratory, personal communication (June 1980).
20. L. F. Hansen, Lawrence Livermore National Laboratory, personal communication (June 1980).
21. D. L. Johnson, P. A. Ombrellaro, and R. E. Schenter, Hanford Engineering Development Laboratory, personal communication (June 1980).

22. W. B. Wilson, T. R. England, R. J. LaBauve, M. Battat, and N. L. Whittermore, "CINDER Actinide Decay Library Development," in "Applied Nuclear Data Research and Development, January 1-March 31, 1980," Los Alamos Scientific Laboratory report LA-8418-PR (1980), p. 28.
23. C. M. Lederer and V. S. Shirley, Eds., Table of Isotopes, Seventh Edition (John Wiley and Sons, Inc., New York, 1978).
24. F. W. Walker, G. J. Kirouac, and F. M. Rourke, Chart of the Nuclides, Twelfth Edition (General Electric Company, Knolls Atomic Power Laboratory, Schenectady, New York, 1977).
25. M. E. Battat, W. B. Wilson, R. J. LaBauve, and T. R. England, "Actinide Decay Data," in "Applied Nuclear Data Research and Development, April 1-June 30, 1980," Los Alamos Scientific Laboratory report LA-8524-MS (1980), p. 18.
26. F. Manero and V. A. Konshin, "Status of the Energy-Dependent  $\bar{\nu}$ -Values for the Heavy Isotopes ( $Z > 90$ ) from Thermal to 15 MeV and of  $\bar{\nu}$ -Values for Spontaneous Fission," At. Energy Rev. 10, 637 (1972).

Printed in the United States of America  
 Available from  
 National Technical Information Service  
 US Department of Commerce  
 5285 Port Royal Road  
 Springfield, VA 22161  
 Microfiche \$3.50 (A01)

Page Range	Domestic Price	NTIS Price Code	Page Range	Domestic Price	NTIS Price Code	Page Range	Domestic Price	NTIS Price Code	Page Range	Domestic Price	NTIS Price Code
001-025	\$ 5.00	A02	151-175	\$11.00	A08	301-325	\$17.00	A14	451-475	\$23.00	A20
026-050	6.00	A03	176-200	12.00	A09	326-350	18.00	A15	476-500	24.00	A21
051-075	7.00	A04	201-225	13.00	A10	351-375	19.00	A16	501-525	25.00	A22
076-100	8.00	A05	226-250	14.00	A11	376-400	20.00	A17	526-550	26.00	A23
101-125	9.00	A06	251-275	15.00	A12	401-425	21.00	A18	551-575	27.00	A24
126-150	10.00	A07	276-300	16.00	A13	426-450	22.00	A19	576-600	28.00	A25
									601-up	†	A99

†Add \$1.00 for each additional 25-page increment or portion thereof from 601 pages up.

ASL  
KODAK SAFETY FILM

100-0 100

RECEIVED